WASHINGTON PUBLIC POWER SUPPLY SYSTEM
P.O. Box 968 • Richland, Washington 99352-0968

September 24, 1997
GO2-97-175

Docket No. 50-397

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21 LICENSEE EVENT REPORT NO. 97-004-01**

Transmitted herewith is revised Licensee Event Report No. 97-004-01 for WNP-2. This report is submitted pursuant to 10 CFR 50.73 and discusses the items of reportability, corrective action taken, and action to preclude recurrence.

Should you have any questions or desire additional information regarding this matter, please call me or P.J. Inserra at (509) 377-4147.

Respectfully,

P.R. ~~Bemis~~
Vice President, Nuclear Operations
Mail Drop PE23

Enclosure

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CATEGORY 1

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ARBUCKLE,J.D.	Washington Public Power Supply System
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RECIP.NAME	RECIPIENT AFFILIATION

SUBJECT: LER 97-004-01:on 970327,plant operators manually scrammed reactor as required TS due to indication of entry into Region A of power-to-flow may.Caused by inadequate attention to detail.Established event evaluation teams.W/970924 ltr.

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	1 of 6

TITLE (4) TECHNICAL SPECIFICATION REQUIRED MANUAL SCRAM DUE TO INDICATION OF ENTRY INTO REGION A OF THE POWER-TO-FLOW MAP

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
03	27	97	97	- 0 0 4 -	0 1	09	24	97	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 4 6

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)

_ 20.402(b)	_ 20.405c	X 50.73(a)(2)(iv)	_ 73.71(b)
_ 20.405(a)(1)(i)	_ 50.36(c)(1)	_ 50.73(a)(2)(v)	_ 73.71(c)
_ 20.405(a)(1)(ii)	_ 50.36(c)(2)	_ 50.73(a)(2)(vii)	_ OTHER (Specify in Abstract below and in Text, NRC Form 366A)
_ 20.405(a)(1)(iii)	_ 50.73(a)(2)(i)	_ 50.73(a)(2)(viii)A	
_ 20.405(a)(1)(iv)	_ 50.73(a)(2)(ii)	_ 50.73(a)(2)(viii)B	
_ 20.405(a)(1)(v)	_ 50.73(a)(2)(iii)	_ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
J.D. Arbuckle, Licensing Technical Specialist	509	377-4601

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (16)

On March 27, 1997 at 0907 hours with the plant in Mode 1 at 46 percent power, plant operators manually scrammed the reactor as required by the Technical Specifications due to indication of entry into Region A of the power-to-flow map during Reactor Recirculation (RRC) and Reactor Feedwater (RFW) System testing. (It was determined by followup engineering analyses that Region A of the power-to-flow map had not been entered.) The post-modification testing was being performed to demonstrate that the feedwater level control system and recirculation flow runback feature would avoid a scram following the trip of a single reactor feedwater pump during power operation.

During the test and after the planned trip of pump RFW-P-1B, an expected reactor recirculation pump speed runback to 27 Hz was observed. Following this planned runback, a second unexpected runback to 15 Hz occurred which, based upon evaluation of plant heat balance data, appeared to have placed the reactor into Region A of the power-to-flow map. The second runback was caused by a reactor recirculation pump differential temperature cavitation interlock condition.

Plant operators responded conservatively by immediately scramming the reactor and maneuvering the plant to a safe shutdown condition in accordance with procedures.

The root cause of this event was inadequate attention to detail and a lack of a questioning attitude during the design phase of the modifications for the adjustable speed drive and digital feedwater level control systems. Corrective actions included the establishment of internal and external event evaluation teams to investigate the event, and successfully repeating the reactor feedwater pump trip test.

There were no indications of power oscillations during the event period. This event posed no threat to the health and safety of the public or plant personnel.

Event Description

On March 27, 1997 at 0907 hours with the plant in Mode 1 at 46 percent power, plant operators manually scrammed the reactor as required by the Technical Specifications due to indication of entry into Region A of the power-to-flow map during Reactor Recirculation (RRC) System [AD] and Reactor Feedwater (RFW) System [JB] testing. (It was determined by followup engineering analyses that Region A of the power-to-flow map had not been entered.)

At the time of the event, post-modification testing of the digital feedwater level control and RRC adjustable speed drive systems was being conducted to demonstrate that the level control system and recirculation flow runback feature would avoid a scram following the trip of a single feedwater pump during power operation.

During the test and after the trip of pump RFW-P-1B [P], a reactor recirculation pump speed runback to 27 Hz was observed as expected. Following this runback, a second unexpected runback to 15 Hz occurred which, based upon evaluation of plant heat balance data, appeared to have placed the reactor into Region A of the power-to-flow map. The second runback to 15 Hz was caused by a reactor recirculation pump differential temperature cavitation interlock condition existing for greater than 15 seconds.

As expected in response to the manual scram, reactor vessel water level decreased to just below +13 inches. In response to the low level condition, reactor feedwater flow increased and level was recovered to above +13 inches. During this evolution and at approximately +18 inches, pump RFW-P-1A [P] speed failed to decrease in response to the increasing level control signal. As a result, reactor vessel water level reached +54 inches (Reactor Vessel Water Level - High: Level 8) and the pump tripped on high level.

Plant operators successfully returned both reactor feedwater pumps to operation and restored level to within normal limits.

Immediate Corrective Action

Plant operators maneuvered the plant to a safe shutdown condition in accordance with procedures.

Further Evaluation

1. Plant operators made a conservative decision to manually scram the reactor when plant heat balance data indicated that Region A of the power-to-flow map had been entered.

   However, based upon a followup review of plant data it was determined by Reactor Engineering personnel that, although close, Region A of the power-to-flow map had not been entered during the event. A review of Average Power Range Monitor [IG] and heat flux data confirmed that conclusion.

2. Based upon the followup conclusion that Region A of the power-to-flow map had not been entered, this event is being reported pursuant to 10 CFR 50.73(a)(2)(iv) as, "Any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System."

   This event was originally determined to be reportable as a four-hour Reactor Protection System actuation in accordance with 10 CFR 50.72(b)(2)(ii). The initial notification to the NRC Operations Center was made on March 27, 1997 at 1013 hours (66 minutes after the event). Upon further review, it was determined that the event should have also been reported within one hour as an initiation of a plant shutdown required by the Technical Specifications in accordance with 10 CFR 50.72(b)(1)(i)(A).

   Technical Specification 3.4.1, "Recirculation Loops Operating," requires that the reactor mode switch be placed in the shutdown position within 15 minutes of operation in Region A of the power-to-flow map. On March 27, 1997 at 1308 hours, the NRC Operations Center was notified of the determination that the event was also reportable within one hour.

   However, since Region A of the power-to-flow map was not actually entered, this event is reportable pursuant to 10 CFR 50.72(b)(2)(ii).

3. Plant design includes an interlock to reduce RRC pump drive flow during conditions which could cause cavitation in the RRC system. A low temperature differential between the RRC pump suction and the vessel steam dome indicates a reduction in subcooling of reactor coolant in the reactor vessel annulus region. A reduction in RRC drive flow under these conditions prevents system cavitation and avoids equipment damage from prolonged vibration.

   Plant design at the time of the event was such that a reduction in drive flow would be initiated if the minimum differential temperature of 9.9 degrees Fahrenheit was exceeded for more than 15 seconds. If this occurred, the control logic would automatically reduce drive speed to 15 Hz, and RRC pump flow would be reduced to 25 percent.

4. Following the planned RRC runback to 27 Hz, evaluation of plant data validated that a differential temperature condition of greater than 9.9 degrees Fahrenheit was present for longer than 15 seconds, and this condition caused the RRC runback to 15 Hz as designed.

5. The differential temperature cavitation interlock was not expected to exist for greater than 15 seconds. The digital feedwater level control and adjustable speed drive systems were previously modeled in the plant simulator and in simulation programs used by General Electric and the Supply System. There was no previous simulator indication that the cavitation interlock would be initiated due to a single reactor feedwater pump trip from 100 percent reactor power.

6. The digital feedwater level control and adjustable speed drive systems performed as designed to the planned trip of the reactor feedwater pump up to the point of the second runback to 15 Hz. Following the manual scram, the digital feedwater level control system did not respond as designed to control vessel level below the Reactor Vessel Water Level - High (Level 8) setpoint of +54 inches.

Root Causes

•Runback The root cause of this event was inadequate attention to detail and lack of a questioning attitude that led to acceptance of previously known design information without in-depth challenge during the design phase of the adjustable speed drive and digital feedwater level control system modifications. The integrated effect of these modifications, in conjunction with implementation of power uprate one year earlier, was not considered in relation to the cavitation interlock protection setpoint. This resulted in failure to identify the potential for activation of the differential temperature cavitation interlock.

•Level-8 The root cause of the failure of the digital feedwater level control system to maintain reactor water less than +54.5 inches (Level-8) was attributed to a design deficiency with the governor controller. This deficiency has existed since the original plant design and it was anticipated that installation of the digital system would correct the problem. Following extensive analyses, testing and intrusive investigations, it was concluded that the only effective means of avoiding vessel overfill was improvement in the responsiveness of the governor controller following a scram. The required changes are being evaluated.

Further Corrective Action

An event evaluation team, composed primarily of Supply System personnel, was established to investigate the event and present the results and conclusions to plant management. Specific areas evaluated by the team included analytical results of testing performed, performance of the on-shift Operations personnel during this event, performance of the digital feedwater level control and adjustable speed drive systems, the adequacy of the test procedure as it relates to the conditions surrounding the scram, and the adequacy of the design related to the installation of the adjustable speed drive and digital feedwater level control systems. As a result of the evaluation, several recommendations were developed. These recommendations have been reviewed and are being implemented as part of the Problem Evaluation Request process.

A second independent evaluation team, consisting of primarily non-Supply System personnel, evaluated the event and performed a critical review of the investigation conducted by the Supply System event evaluation team. The independent team validated the findings of the event evaluation team.

TEXT (17)

The cavitation interlock setpoint and methodology were reviewed to determine if any runback value or cavitation logic changes were necessary. As a result of this review, the differential temperature cavitation interlock setpoint was changed from 9.9 degrees Fahrenheit to 10.7 degrees Fahrenheit. In addition, the differential time delay to runback was increased from 15 seconds to ten minutes.

To demonstrate that the feedwater level control system and recirculation flow runback feature would prevent a scram following the trip of a single feedwater pump during power operation, the test was successfully repeated on July 23, 1997 when the plant returned to stable, full power operation following the R-12 Maintenance and Refueling Outage. The systems responded as designed.

An assessment of the Level-8 design function was performed. Based on this assessment, it was concluded from the evaluation that a post-scram, Level-8 trip of this type does not represent a safety concern.

Assessment of Safety Consequences

The probability of thermal-hydraulic oscillations is greatly increased if the plant is operating in Region A of the power-to-flow map. Plant operators made a conservative decision to manually scram the reactor when plant heat balance data indicated that Region A of the power-to-flow map had been entered. This action was taken well within the 15 minute time-frame allowed by the Technical Specifications. Subsequent operator actions were prompt and correct to maneuver the plant to a safe shutdown condition.

Upon further evaluation it was determined that, although close, the plant had not actually entered Region A of the power-to-flow map. There were no indications of power oscillations during the event period. Accordingly, the event did not represent a safety concern.

An evaluation was performed to determine the impact of continued operation of the plant without a final design resolution of this problem. It was concluded from the evaluation that a post-scram, Level-8 trip of this type does not represent a safety concern. The post scram response is not dissimilar to what would have been seen with the previous analog system. Since the Level-8 trip was reached post-scram, there was no adverse impact on the fuel thermal limits. For long-term cooling and inventory make-up, the High Pressure Core Spray [BG] and Reactor Core Isolation Cooling [BN] Systems would be available once the water level lowered to their initiation setpoints. Therefore, the transients in the FSAR are still bounding and, as a result, the consequences of an accident as analyzed in the FSAR were not increased.

Although additional operator actions may be required to preclude or recover from a Level-8 trip, plant operators are trained that a Level-8 trip following a scram may occur. This training includes actions that can be taken to either avoid the trip or respond accordingly should a trip occur.

TEXT (17)

Similar Events

Licensee Event Report 96-004 reported a problem involving a manual reactor scram due to a reactor water level transient during testing of the digital feedwater system. After performing a step of the test procedure which reduced reactor water level by six inches, level continued to decrease past the intended value. The feedwater level control system responded to control vessel level below the Level-8 setpoint during this event (a Level-8 trip signal was not received).

The cause of this event was determined to be a manufacturer's programming error in the digital feedwater system testing software which caused mismatches to occur in the feed/steam signal, resulting in unwanted level step changes. The software was modified by the vendor to delete the possibility of unwanted step changes. The software change was verified and validated and functionally tested on the plant simulator prior to installation in the plant.

Corrective action taken in response to the previous event would not have been expected to preclude this event. The action was designed to correct a specific software problem that was causing mismatches in the feed/steam signal.